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PG&E Letter DCL-03-080

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding "License Amendment Request 03-03, Revision to Technical Specification 3.5.2 – Increase in Completion Time for Charging Pump During Unit 1 Cycle 12 from 72 Hours to 7 Days"

Dear Commissioners and Staff:

In Pacific Gas & Electric (PG&E) letter DCL-03-019, "License Amendment Request 03-03, Revision to Technical Specification 3.5.2 – Increase in Completion Time for Charging Pump During Unit 1 Cycle 12 from 72 Hours to 7 Days," dated February 28, 2003, PG&E requested NRC approval of a one-time change to Technical Specification 3.5.2, "ECCS – Operating," Action A, to increase the completion time for Unit 1 Centrifugal Charging Pump 1-1 during Unit 1, Cycle 12, from 72 hours to 7 days.

In a telephone call held between the NRC staff and PG&E personnel on June 5, 2003, the NRC staff requested additional information concerning PG&E DCL-03-019.

The requested information and PG&E's responses are provided in Enclosure 1.

This additional information does not affect the results of the technical evaluation and no significant hazards consideration determination previously transmitted in PG&E letter DCL-03-019.

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If you have any questions or require additional information, please contact Stan Ketelsen at (805) 545-4720 or Tom Grozan at (805) 545-4231.

Sincerely

David H. Oatley
Vice President and General Manager -Diablo Canyon

jer/3664
Enclosures

cc:	Edgar Bailey, DHS
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	)	Docket No. 50-275 Facility Operating License No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	)	Docket No. 50-323 Facility Operating License No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath states that he is Vice President and General Manager - Diablo Canyon of Pacific Gas and Electric Company; that he has executed this response to the NRC request for additional information on License Amendment Request 03-03 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

DH Oatley

David H. Oatley
Vice President and General Manager - Diablo Canyon

Subscribed and sworn to before me this 26th day of June, 2003.

Chuck Mackey

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1397547
Notary Public - California
San Luis Obispo County
My Comm. Expires Feb 1, 2007

NRC Request No. 1

Discuss the current reliability data for the centrifugal charging pumps (CCPs) including any adverse trends.

PG&E Response:

PG&E has reviewed the NRC performance indicators for high pressure safety injection and maintenance rule unavailability for high head injection. NRC performance indicators for the Diablo Canyon Power Plant (DCPP) Units 1 & 2 show a flat or slightly decreasing trend that is well below the Increased Regulatory Response Band Threshold (1.5 percent - PWR HPSI) stated in NEI 99-02, Revision 2, Table 1. The Unit 1 maintenance rule unavailability is trending flat, well below the performance criteria. The Unit 2 maintenance rule unavailability is trending slightly down and is also well below the performance criteria. PG&E knows of no issues that would impact these trends other than the seal leakage of CCP 1-1. Therefore, PG&E concludes there are no adverse trends associated with the CCPs.

NRC Request No. 2

Confirm that the PRA calculation for LAR 03-03 did include internal events, including flooding, and seismic events, but did not include fire events. Explain why it was not necessary to include fire events.

PG&E Response

Calculation PRA03-03 evaluated the "combined" core damage frequency (CDF) and large early release frequency. "Combined" is internal CDF (including internal flooding) and seismic CDF. It does not include any contribution from the DCPP fire vulnerability probabilistic risk assessment (PRA). The fire PRA was reviewed and it was determined that a single CCP out of service would not noticeably affect the resulting PRA. It should be noted that CCP 1-1 and CCP 1-2 (the redundant CCP) are located in the same fire area. The scenario analyzed in the PRA is the loss of both CCPs due to fire, and the resulting CDF is 2.90E-9/year. Because both of the CCPs are in the same fire area, DCPP credits the positive displacement charging pump, which is not modeled in the PRA, for the Appendix R safe shutdown requirements.

NRC Request No. 3

Provide a copy of the Executive Summary from Calculation File C.9, Revision 9, dated March 15, 2001, that describes the DCPP PRA model.

PG&E Response

A copy of the executive summary Attachment A.1 from Calculation C.9, Revision 9, including some minor corrections that do not effect License Amendment Request (LAR) 03-03, is included in Enclosure 2.

NRC Request No. 4

Along with Item 3, provide a summary of the major updates made from the PRA DC1997 Model (used as the basis for the previous LAR 00-04) to the DCC0 Models (used as the basis for LAR 03-03). Confirm that there are no outstanding Peer Review comments that would have any noticeable affect on the PRA calculation for LAR 03-03.

PG&E Response

The following is a summary of major changes made to the model between the DC1997 and DCC0 models. A more detailed description of the model changes is provided in Section 2.0 of Enclosure 2.

1. Initiating event frequencies were updated from a previous contractor supplied database (early 1980s) to the newer frequencies identified in NUREG-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants 1987-1995," dated February 1999.
2. Electric power capability was updated to include the second switchyard (500 kV) in addition to the 230 kV switchyard. Also the capability to cross-tie one class 1 diesel generator (DG) to another vital bus upon failure of the DG on that bus was added.
3. The human reliability analysis was upgraded to complete all the type "A" observations from the Peer Review.
4. System analyses were updated for the anticipated transients without scram mitigation system actuation circuitry, the solid state protection system, the auxiliary saltwater system, and the safety injection system.

PG&E has reviewed the outstanding Peer Review comments and has determined that there are no comments that would have a noticeable effect on the results of the PRA analysis. The conclusions would remain the same.

Attachment A.1

EXECUTIVE SUMMARY from Calculation File C.9, Revision 9, dated March 15, 2001

Attachment A.1

EXECUTIVE SUMMARY

1.0 INTRODUCTION

This document provides an executive summary of the Diablo Canyon Power Plant (DCPP) Probabilistic Risk Assessment (PRA) model.

The analyses summarized in this report represent an enhancement of the original Diablo Canyon Probabilistic Risk Assessment (DCPRA-1988) (Reference A-1) performed as part of the Long Term Seismic Program (LTSP) (Reference A-2). The LTSP re-evaluated the seismic design bases for DCPP, as specified in the Unit 1 Full-Power Operating License, DPR-80, Condition 2.C.(7). As part of the LTSP, PG&E was required by Element 4 of the license condition to complete "a probabilistic risk analysis and deterministic studies, as necessary, to assure adequacy of seismic margins." To meet this requirement, the DCPRA-1988 was completed in 1988 by a team of consultants and utility personnel lead by PLG, Inc. (PLG). The DCPRA-1988 is a full-scope Level 1 PRA that evaluated the probable frequency of experiencing reactor and plant damage resulting from internal and external initiating events. While it was performed for DCPP Unit 1, the DCPRA-1988 is equally applicable to DCPP Unit 2 because of the substantial similarities between the two units. The Nuclear Regulatory Commission (NRC) reviewed the LTSP and issued Supplement No. 34 to NUREG-0675 (Reference A-3) in June 1991, accepting the DCPRA-1988. Brookhaven National Laboratory (BNL) performed the primary review of the DCPRA-1988 for the NRC; their review is documented in NUREG/CR-5726 (Reference A-4). In addition, the Advisory Committee on Reactor Safeguards accepted the NRC's review of the LTSP and DCPRA-1988 and concluded that DCPP "can be operated without undue risk to the health and safety of the public" (Reference A-5).

To fulfill the requirements of NRC Generic Letter 88-20 (Reference A-6), and as part of PG&E's ongoing risk management tasks, PG&E updated the DCPRA-1988 to:

- Further enhance PG&E's familiarity with all aspects of the PRA. PG&E assumed primary responsibility for the PRA model. This required conversion of the PRA models to PG&E in-house computers using PLG software.

- Reflect current plant design and operation. This included incorporation of updated design and operational data through June 1990 and December 1989, respectively. In addition, PG&E incorporated PLG, PG&E, and NRC/BNL comments into the PRA model. PG&E also maintains a "living PRA" with periodic updates.

- Expand the DCPRA-1988 to include the Level 2 containment performance analysis.

- Develop a risk management program. Using the insights and knowledge gained from the development of the DCPRA-1988 and its subsequent update, PG&E committed to addressing the impact on risk resulting from plant maintenance activities, equipment modifications, and operator actions, as appropriate.

The update of the DCPRA-1988 model is referred to as the DCPRA-1991 model. In 1993, the model was further updated during the Individual Plant Examination of External Events (IPEEE) study. The significant enhancements in 1993 included an update of the external events initiators and evaluations, as well as an update of the emergency diesel generator (EDG) configuration. The change in EDG configuration was necessary to reflect installation of the third Unit 2 EDG, the resulting configuration being 3 EDGs per unit.

2.0 LIVING PRA

As part of maintaining a living PRA, the model has been periodically updated, as described below.

DCPRA-1988 Model – As stated above, this was the original model developed for the LTSP as part of a condition of licensing DCPP. Both PG&E personnel and its consultants (mainly PLG) developed the model, which resided on PLG's mainframe computer system. The model was reviewed by the NRC staff and its consultant, BNL (documented in References A-1 through A-5).

Core Damage Frequency:

- Internal 1.30E-4
- Seismic 3.7E-5
- Fire 3.9E-5
- Total 2.04E-4

DC1990 Model – This was a transition model and was used to migrate the model from PLG's mainframe computer system to RISKMAN on the PC and compare the results with the DCPRA 1988 model. This update began model documentation under PG&E's procedures, which required verification of each calculation file by a second person.

DC1991 Model – This model was used to support PG&E's response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities" (the DCPP IPE). Internal flooding and containment performance (Level 2) were added to the model at this time. The PRA database for plant design and operational data was updated through December 31, 1989.

Core Damage Frequency

- Internal 8.8E-5
- Total 1.55E-4

Major items of the update were:

- firewater cooling to charging pumps
- DFO pump recirculation path modifications
- Numerous model changes based on BNL feedback (Ref. A-6 Table 5-4)

The NRC documented their evaluation of the DCPP IPE in a letter dated June 30, 1993, which included a review of the Level 2 by its contractor, Scientech, Inc. (SCIE-NRC-210-92).

DC1993 Model – This model was used to support PG&E's response to Generic Letter 88-20, Supplement 4, "Individual Plant Response to External Events for Severe Accident Vulnerabilities," (the DCPP IPEEE). The PRA database for plant design and operational data was updated through December 31, 1991. The Seismic and Fire PRA models from the DCPRA were updated, along with the internal model.

Core Damage Frequency

- Internal 5.52E-5
- Seismic 4.0E-5
- Fire 2.8E-5
- Total 1.23E-4

Major items of the update were:

- Sixth diesel generator was added

The NRC documented their evaluation of the DCPP IPEEE in a letter dated December 4, 1997, which included a Step 1 review of the IPEEE by its contractor, Energy Research, Inc. (ERI), ERI/NRC 95-503.

DC1995 Model -- The main objectives of this update were to incorporate into the model plant-specific information on component reliability and unavailability data through December 31, 1994. Additionally, the model was updated based on pertinent and important plant hardware and procedural changes. Appropriate industry events and PRA staff comments and observations about the previous model were incorporated. This was the first model used to support online risk assessments for the Operations department.

Core Damage Frequency

- Internal 4.52E-5
- Total 1.13E-4

Major items of the update were:

- Upgraded the Class 1 instrument inverters
- Installed high temperature qualified "O" rings in the RCP seals
- Added the backup Class 1 battery chargers

DC1997 Model -- During this update, two auxiliary tasks were performed in addition to the update activities referred to for the DCPRA-1995 model. These tasks involved the creation of a Large Early Release Frequency (LERF) model and re-evaluation of the Control Room fire scenarios. The online risk configurations were further refined to support the online risk assessments performed by the Operations department. The PRA database for plant design and operational data was updated through September 30, 1996. The internal section of this model was used for risk information in support of the one time AOT extension for Centrifugal Charging Pump 2-1 submitted June 2, 2000 in PG&E Letter DCL-00-086.

Core Damage Frequency

- Internal 3.32E-5
- Seismic 3.7E-5
- Fire 2.7E-5
- Total 9.72E-5

Major items of the update were:

- Creation of a LERF model
- Reevaluation of the Control Room fire scenarios based on more recent fire PRA methodologies
- Reevaluated PTS data up to 2005
- Added backup water supply to the AFW pumps
- Increased SSPS testing intervals to reflect recent changes in Technical Specifications
- Increased the level of detail in the ASW modeling

DC00 Model -- This update (completed May 2000) incorporated substantial changes to two system calculations, Auxiliary Saltwater and Pressure Relief. The model was also expanded to include Balance of Plant systems. This was a combined model, but it was only quantified for internal events.

Core Damage Frequency

- Internal 1.4E-5
- Total 7.8E-5

Major items of the update were:

- Changed ASW success criteria to be consistent with the SBO thermal hydraulic analysis
- Added the third PORV
- Added BOP systems (feedwater, condensate, circulating water/service water, non-vital power, and completed modeling of instrument air)
- Updated Common Cause Failure (CCF) model (Alpha Factors and CC database variables) for selected components
- Re-evaluated initiating event frequencies based on more recent industry data (NUREG-5750)

This model was used for the Westinghouse Owners' Group Peer Certification visit in May 2000.

DC00ISI Model -- This model (completed June 2000) was developed to quantify the DC00 combined model for the Risk-Informed In-service Inspection (RI-ISI) application. This was a combined model, but it was only quantified for internal and seismic events, and the results were only evaluated for their ability to support the RI-ISI application.

Core Damage Frequency

- Internal 1.4E-5
- Seismic 3.4E-5
- Fire 1.5E-5
- Total 6.3E-5

DCC0 Model -- This model (completed March 2001) builds on all the items above, and quantifies the combined model; i.e., internal, seismic, and fire initiating events. This model incorporated all the Type "A" comments from the WOG Peer Review performed in May 2000, and will be released for risk assessments by the Operations department and other uses.

Core Damage Frequency

- Internal 1.04E-5
- Seismic 3.12E-5
- Fire 1.33E-5
- Total 5.5E-5

Major items of the update were:

- Revised the pre-initiator HRA as a result of items in the Peer Certification
- Included new and revised human reliability analysis (HRA) for post initiators
- Added the 500kV switchyard to the model and took credit for the ability to feed more than one vital bus from the Class 1 EDGs
- Refined SSPS modeling by removing some conservatisms identified during the Peer Review, including steam line break inside containment modeling
- Added AMSAC to the model as a diverse start of AFW pumps and turbine trip

3.0 PLANT FAMILIARIZATION

DCPP is located on California's central coast in San Luis Obispo County, approximately 12 miles west-southwest of the city of San Luis Obispo. The plant consists of two separate but essentially identical units (Units 1 and 2).

Each unit was built with a four-loop pressurized water reactor (PWR) nuclear steam supply system (NSSS) furnished by Westinghouse Electric Corporation. The NSSS for each unit is contained within a steel-lined, reinforced concrete structure that is capable of withstanding the pressure that might be developed as a result of the most severe design basis loss of coolant accident (LOCA).

Although the reactors, structures, and all auxiliary equipment are essentially identical for the two units, there was initially a difference in the thermal power capacities of the reactors. The original licensed reactor rating for Unit 1 was 3338 MWt, with a corresponding net maximum dependable capacity rating of 1073. In November 2000, Unit 1 thermal power was up-rated to 3411 MWt. The corresponding net maximum dependable capacity rating is still under evaluation. Unit 2 is (and has always been) rated for 3411 MWt, with a corresponding net maximum dependable capacity rating of 1087 MWe.

Unit 1 received its full power operating license from the NRC on November 2, 1984, and began commercial operation on May 7, 1985. Unit 2 received its full power operating license on August 26, 1985, and began commercial operation on March 13, 1986.

A detailed description of the plant site, facilities, and safety criteria is documented in the FSAR Update, as well as in the DCPRA-1988 (Reference A.1).

4.0 OVERALL METHODOLOGY

The DCPRA model methodology closely follows the analytical tasks and methods developed by PLG, which were implemented in performing more than 20 full-scope and phased PRAs of U.S. and foreign nuclear power plants. The original exposition of the theoretical and mathematical bases for the approach is given in the PLG methodology document (Reference A.7).

5.0 SUMMARY OF RESULTS

This section summarizes the results of the current PRA Model. In general, more significant figures are presented in the results tables than the accuracy of the PRA provides. However, the additional significant figures are important for sensitivity studies and allow for traceability and reproducibility of results.

5.1 Core Damage Frequency Results

In the DCC0 combined model, the CDF and the LERF figures of merit results are as follows:

Model	CDF	LERF
Internal	1.04E-5	4.94E-7
Seismic	3.12E-5	1.28E-6
Fire	1.33E-5	6.31E-9

Since there are uncertainties in the initiating event frequencies, component failure rates, and equipment maintenance unavailability, the uncertainty in the CDF and LERF figures of merit are also analyzed but are not presented here. Refer to PRA Calculation C.10.

Table 1 lists the contributions of the major initiating event groups to CDF and LERF. It can be seen that loss of offsite power, loss of ASW or CCW, and transient events combine to contribute approximately 64% of the total CDF. From the CDF point of view, the loss of offsite power event is important because of the challenge it places on the plant, combined with the moderate frequency of the initiator. In particular, a loss of offsite power challenges the emergency diesel generators in the plant. Consequently, many of the core damage sequences are predicated on an unavailability of one or more diesel generators, causing the loss of vital electric power trains. The lack of one or more trains of vital electric power can place safety-related systems in degraded states and increase the likelihood of their failure.

The loss of ASW and CCW initiating events are important because, even though of low frequency, they would result in a significant degradation of the RCP seal cooling function. Both thermal barrier cooling and seal injection functions would be lost unless an alternate cooling method is provided for the charging pumps, as well as the loss of safety injection function (i.e., a LOCA with no RCS makeup capability). This significant consequence is due to the dependency of the safety injection function on the ASW/CCW systems.

The general transients such as reactor trip and turbine trip are mild transients, but occur relatively frequently, which cause them to contribute 17% of the CDF. The design basis accidents, such as medium and large LOCAs, have a modest contribution to the CDF because they are unlikely events, i.e., the initiator frequency is relatively low.

From Table 1 it can also be seen that non-isolable SGTR and Interfacing System LOCAs combine to contribute approximately 98% of the total LERF. The relatively large contribution from these initiators is due to the capability of an isolated containment to withstand the consequences of all core damage sequences.

Table 2 presents the major core damage accident types based on the similarity of accident sequences. With the exception of the ATWT events, these accident types were quantified using group reports and setting the appropriate top events to failure (e.g., PRX, SE, OB, and VI). One of the major core damage accident types is the transient-induced LOCA occurring through a stuck-open pressurizer PORV (~12%). This failure scenario is also strongly influenced by the loss of offsite power initiating event. A loss of offsite power results in a challenge to the pressurizer PORVs and a degradation of electric power support to the PORV block valves, if one or more diesel generators fail. If a pressurizer PORV fails to reclose, high pressure ECCS injection (centrifugal charging or safety injection) must continue to maintain RCS level to prevent core uncovery. The potential reduction in available electric power trains, however, can place the ECCS equipment in a degraded state of operation.

5.1.1 Top Event Importance

Table 3 provides an importance ranking (based on the F-V importance measure) of key top events for the CDF figure of merit.

The dominance of RCP seal LOCAs is illustrated by the importance of the loss of RCP seal integrity and the unavailability of the auxiliary saltwater system.

The significance of the loss of offsite power initiating event is reflected in the importance of the emergency diesel generators. DCPP has three dedicated diesel generators per unit. The importance of RCS pressure relief and pressurizer PORV reclosure also reflects the dominance of the loss of offsite power initiating event. As discussed earlier, the loss of offsite power initiating event has the potential to not only challenge the pressurizer PORVs, but to also degrade the electric power supporting the PORV block valves if one or more diesel generators fail to operate.

Table 4 provides an importance ranking (based on the F-V importance measure) of key top events for the LERF figure of merit. The importance of non-isolable SGTR events and large interfacing system LOCA events is clearly shown by Top Events SL (failure to isolate SGTR) and SM (large leak rates). The importance of operator actions is reflected in Top Event MU (makeup to the RWST) and OP (operator secures SI during SGTR). Other events result in the primary side at high pressure and the secondary side dry (high and dry sequences).

5.1.2 Important Operator Actions

Operator actions performed before and during an accident play a significant role in reducing the frequency of core damage. Important operator actions are discussed below.

- Reduce Unnecessary Component Cooling Water (CCW) Loads

  If the temperature of the CCW system increases above its allowable limits, the operators will attempt to reduce unnecessary loads on CCW header C.

- Backup Cooling for Centrifugal Charging Pumps

  If the CCW system is failed, RCP seal injection can be maintained via a centrifugal charging pump (which normally requires CCW cooling) with operator action to align firewater to provide cooling to the centrifugal charging pump.

- 480V Switchgear Ventilation Recovery

  Without operator intervention, failure of the 480V switchgear ventilation system is assumed to cause high room temperatures that lead to the loss of all Class 1E 480V AC, the battery chargers, the inverters, and eventually all 125V DC when the batteries drain down. The operators have approximately six hours to open the switchgear and inverter/battery charger room doors and install alternate means of ventilation before the failure of the 480V switchgear ventilation system would lead to a plant trip. If the plant trips, it is assumed that one and possibly two inverters have failed. The operators can still recover from the transient by opening doors to the inverter/battery charger rooms and selecting backup power for any failed inverter within two hours.

- Electric Power Recovery

  During full or partial station blackout scenarios, the PRA assumes it is important that the operators: (1) restore AC power to at least two vital buses by restoring offsite power, recovering diesel generators, or crosstying vital buses; (2) maintain and control auxiliary feedwater flow from an AFW pump; (3) monitor core subcooling and reactor coolant inventory; and (4) monitor DC power availability and take action to extend battery life.

- Switchover to Cold Leg Recirculation Mode

  During small, medium, and large LOCAs, operator actions are necessary to switchover from the safety injection mode to cold leg recirculation. The actions required for successful switchover are involved and the allowable completion time varies depending on the event (within five hours after a small LOCA, one hour after a large LOCA).

- Initiation of Bleed and Feed Cooling

  If AFW is unavailable, decay heat can be removed from the RCS through bleed and feed cooling (feeding with the charging and safety injection pumps, bleeding through the pressurizer PORVs) in accordance with operator procedures.

- Isolation of Ruptured Steam Generator

5.2 Containment Performance Results

As stated earlier, the 1991 update developed a Level 2 model to address the consequences of core damage. The Level 2 results were grouped into the release categories discussed below.

Cat 1 - Small, Early Containment Failure

The small, early containment failure release group is defined as occurring within four hours of vessel breach and is a containment failure equivalent to three inches or less in diameter.

Cat 2 - Large, Early Containment Failure

The large, early containment failure release group is defined as occurring within four hours of vessel breach and is a containment failure equivalent to or greater than three inches in diameter. The major causes of large, early containment failures are summarized in the following table.

Large, Early Containment Release Cause
High Pressure Melt Ejection
Hydrogen Burns within 4 Hours of Vessel Breach
In-Vessel Steam Explosions
Hydrogen Burns at Vessel Breach
Containment Isolation Failure

Cat 3 - Late Containment Failure

Late releases occur more than four hours after vessel breach. This group includes both large and small containment failures. A transient that severely degrades the ECCS and is complicated by an RCP seal LOCA or stuck-open PORV initiates the sequences most likely to be in this release category.

Cat 4 - Containment Bypasses

The containment bypass release group provides the most significant releases and includes sequences involving interfacing system LOCAs and non-isolable SGTRs. SGTRs are divided into two types: initiating events and those that are induced. Initiating event SGTRs (unisolated cases) are assumed to involve a single tube that results in a smaller release than for an interfacing system LOCA. Induced SGTRs are assumed to involve several tubes and have releases similar to a large interfacing system LOCA.

During the 1997 update, based on common industry practice, it was judged that only the large early release category would be of significant interest in the current risk-informed environment. Therefore, based on an assessment of the categories presented above, a LERF model was developed that concluded the major contributors to the LERF figure of merit are all the containment bypass contributors and core damage sequences which coincide with the large containment isolation failure.

Because DCPP has a large, dry PWR containment, which plant-specific analysis determined to be robust, no other containment bypass sequences have been identified that would contribute to the LERF figure of merit. Additionally, industry evaluations have not identified any failure mode vulnerabilities for similar containments. Therefore, it is judged that a simplified LERF model (developed based on the Level 1 results) is sufficient for use in the risk-informed applications. However, the Level 2 model has not been disbanded, and industry events and studies are monitored to ensure that any new issues affecting this judgment are identified and adequately addressed.

5.3 Risk Insights

This section summarizes the risk insights. As recommended in Reference A.8, risk insights were incorporated during development of the DCPP Severe Accident Management Guidelines.

5.3.1 Core Damage Risk Insights

The results indicate that the failure of RCP seal cooling constitutes a significant percentage (~63%) of the total CDF. While this does not qualify as a vulnerability, methods to reduce the frequency of RCP seal LOCAs are being investigated at PG&E and within the industry.

The failure of RCP seal cooling is influenced by the availability of the CCW system. As discussed above, even with the loss of CCW, RCP seal cooling can be maintained via a centrifugal charging pump by aligning an alternate cooling water source (firewater) to the centrifugal charging pump. Nevertheless, the PRA predicts that RCP seal cooling failures (leading to RCP seal LOCAs) are a large contributor to core damage due to the importance of the CCW system.

The PRA success criteria for the CCW system are assumed to be two of three CCW pumps and one CCW heat exchanger, unless unnecessary loads to CCW header C are reduced. The PRA shows CCW system failure is dominated by failure of the operator action to reduce loads on header C. In addition, the PRA success criteria of two CCW pumps (without reducing loads on header C) may be overly conservative for many sequences, although no detailed CCW deterministic analysis has been performed.

Another large contributor to CDF is the transient-induced LOCA (~12%) occurring through a stuck-open pressurizer PORV. Stuck-open PORV core damage scenarios are predominately initiated by loss of offsite power events and typically include failed diesel generators, resulting in the inability to close the PORV block valves without cross tying vital buses.

5.3.2 LERF Risk Insights

The results indicate that non-isolated SGTRs and interfacing system LOCAs constitute approximately 98% of the LERF contributions (66% and 32%, respectively). The frequency of these two initiators is 4.84E-7 per year. The guidelines in Reference A.8 state that no action is necessary for core damage and containment bypass sequences with a frequency this low.

Steam generator tube rupture core damage sequences contributing to the LERF figure of merit are dominated by containment bypass due to stuck-open steam generator atmospheric dump valves or safety relief valves. Steam generator tube rupture core damage sequences with containment bypass caused by operator failure to terminate ECCS injection (overfill) also have a significant contribution to the LERF figure of merit.

6.0 REFERENCES

A.1. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific Gas and Electric Company, PLG-0637, July 1988

A.2. Pacific Gas and Electric Company, "Long Term Seismic Program Final Report," PG&E Letter No. DCL-88-192, July 31, 1988

A.3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report," NUREG-0675, Supplement No. 34, Docket Nos. 50-275 and 50-323, June 1991

A.4. Bozoki, G., et al., "Review of the Diablo Canyon Probabilistic Risk Assessment," NUREG/CR-5726 (DRAFT), June, 1991

A.5. Pacific Gas and Electric Company, "Results of the Advisory Committee on Reactor Safeguards Meeting on the Diablo Canyon Long Term Seismic Program," PG&E Letter No. LTSP 1.3.1, 1.3.2, Log 91-413, Chron. No. 179295, October 21, 1991

A.6. U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, November 23, 1988

A.7. S. Kaplan, G. Apostolakis, B.J. Garrick, D.C. Bley, and K. Woodard, "Methodology for Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0209, June 1981

A.8. NUMARC, "Transmittal of NUMARC Severe Accident Issue Closure Guidelines," NUMARC Report 91-04, February 21, 1992

Table 1 Contributors to Core Damage and Large Early Release Frequency				
Initiating Event Category	Point Estimate CDF (Per Year)	CDF Percent Contribution	Point Estimate LERF (Per Year)	LERF Percent Contribution
Loss of Offsite Power	1.85E-6	18	2.38E-9	<1
General Transients (Reactor Trip, Turbine Trip, Total Loss of Main Feedwater, ATWT, etc.)	1.75E-6	17	4.12E-9	<1
LOCAs (Excessive, Large, Medium, Small, RCP Seal)	1.16E-6	11	7.69E-10	<1
Loss of One 125V DC Bus (F, G, or H)	1.74E-7	1	1.49E-9	<1
Loss of ASW or CCW	2.44E-6	24	0	0
Floods	2.34E-6	23	1.45E-11	<1
Loss of Ventilation (480V Switchgear or Control Room)	4.46E-8	<1	4.59E-10	<1
Steam Generator Tube Rupture	3.85E-7	4	3.24E-7	66
Interfacing System LOCAs	1.60E-7	2	1.60E-7	32
Loss of Instrument AC	1.03E-11	<1	0	0
Steam Line Break	8.35E-8	<1	9.99E-10	<1
Total	1.04E-5	100	4.94E-7	100

Table 2 Major Core Damage Accident Types	
Accident Types	**Point Estimate CDF (Per Year)**
Transient-Induced LOCA (Top Event PRX=F)	1.2E-6
Non-Station Blackout RCP Seal LOCA (Top Event SE=F)	6.5E-6
Bleed and Feed (Top Event OB=F)	4.8E-7
Pressurized Thermal Shock (Top Event VI=F)	8.1E-8
ATWT Events (All 9 ATWT Initiators)	7.4E-8

Table 3 Top Event Importance Ranking F-V for CDF		
F-V Ranking	**Top Event Failure Description**	**F-V**
1	Top Event SE – Reactor Coolant Pump Seal Integrity	.434
2	Top Event GF – Emergency Diesel Generator 1-3 (Bus F)	.195
3	Top Event GG — Emergency Diesel Generator 1-2 (Bus G)	.174
4	Top Event PRX – Transient-Induced LOCA (PORV)	.115
5	Top Event GH – Emergency Diesel Generator 1-1 (Bus H)	.106
6	Top Event AS – Auxiliary Saltwater System	.083
7	Top Event OG –230kV System	.073
8	Top Event CC – Component Cooling Water System	.070
9	Top Event RE – Electric Power Recovery	.064
10	Top Event MU – Makeup to RWST	.062

Table 4 Top Event Importance Ranking F-V for LERF		
F-V Ranking	**Top Event Failure Description**	**F-V**
1	Top Event SL – Isolation of Ruptured Steam Generator	.416
2	Top Event MU – Makeup to RWST	.391
3	Top Event SM – Large ISLOCA	.297
4	Top Event OP – Operator Secures SI During SGTR	.231
5	Top Event LA – RHR Train A	.124
6	Top Event LB – RHR Train B	.124
7	Top Event AH – Bus H Vital AC Power	.063
8	Top Event AG – Bus G Vital AC Power	.058
9	Top Event AW – Auxiliary Feedwater System	.026
10	Top Event LR – Large Early Release Categorization	.023